ANALYSIS OF IRRADIATION MATRIX FOR THE JAPANESE FeCrAl-ODS TEST FUEL RODS IRRADIATIONS AT THE HALDEN REACTOR USING FEMAXI-7 CODE

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ABSTRACT

In Japan, Oxide Dispersion Strengthened (ODS) type of FeCrAl steel (FeCrAl-ODS steel) is being developed for boiling water reactors (BWRs). The key parameters and conditions for planning test irradiations for validating fuel performance analysis of FeCrAl-ODS steel cladded fuel rod under Boiling Water Reactor (BWR) operating condition are presented. Assuming low irradiation creep rate of the cladding, three test cases have been identified. Namely, (1) Design Target Case, (2) High PCMI Case, (3) Cladding Creep-Down Case. The key design parameters are linear heat rate, gap size, filler gas compositions and pressure. The differences among the three cases are expected to be well captured through online measurements of the pellet centerline temperature, rod internal pressure, cladding outer diameter change and elongation. However, significant cladding creep down may not be captured under BWR normal operation condition, if the irradiation creep characteristics of FeCrAl-ODS steel is comparable to that of SUS304.

1. Introduction

Following the Fukushima Daiichi reactors accidents, Accident Tolerant Fuel (ATF) is being developed with different candidate cladding materials for enhanced tolerance to high temperature steam under loss of cooling accident conditions [1]. Among several candidate claddings, FeCrAl steels are promising for its low reaction rate with steam at elevated temperature [2]. In Japan, Oxide Dispersion Strengthened (ODS) type of FeCrAl steel (FeCrAl-ODS steel) is being developed for boiling water reactors (BWRs), as it has excellent creep strength, which allows reduction in the cladding wall thickness to compensate for the neutron penalty relative to Zircaloy [3] – [6]. The project is funded by Ministry of Economy, Trade and Industry (METI) of Japan. In brief, the project covers acquisition and accumulation of manufacturing experience and material properties (welding and fabrication, thermal and mechanical properties, behaviors at normal operation condition and accident conditions) [7-10], evaluation of reactivity and thermal characteristics of the core and the fuel rod [11, 12], and evaluation of impacts on severe accident scenarios [13].

In the preceding study, features of fuel performance of the ATF with FeCrAl-ODS steel cladding under BWR operation condition was evaluated with FEMAXI-7 fuel performance analysis code [12]. In brief, the study indicated that the thermal and mechanical behaviors of the FeCrAl-ODS cladded fuel rod with the cladding wall thickness of 0.30 mm was equivalent to those of typical BWR 9 by 9 type fuel rod with Zircaloy cladding. However, the study also indicated reliable evaluation of the fuel – cladding gap size and Pellet-Cladding Mechanical Interaction (PCMI) with consideration of irradiation creep characteristics of FeCrAl-ODS steel was necessary.

Hence, it is necessary to validate fuel performance analysis codes, such as FEMAXI-7, with irradiation data of FeCrAl-ODS steel cladded fuel under simulated BWR operation condition. However, such irradiation data are currently not available. Moreover, unlike test irradiations of Zircaloy cladded fuel rod, anticipating appropriate irradiation test conditions for such code validation is difficult, because of lack of irradiation experience with the new cladding material. As indicated by the preceding study, the key feature of FeCrAl-ODS steel is its low creep strain rate, which is expected to have significant influence on the fuel – cladding gap conductance and PCMI. This study aims to clarify the key parameters and conditions for planning such test irradiations and propose a test irradiation matrix through analyses of FeCrAl-ODS cladded fuel rod with FEMAXI-7 code. The IFE Halden test reactor is considered as the reference.

2. FEMAXI-7 models

2.1 Calculation geometry and basic method

FEMAXI-7 is one of the representative fuel performance evaluation codes for Light Water Reactors with deterministic method [14]. The code considers a single fuel rod in an axis-symmetric cylindrical geometry as shown in **Fig. 1**. Fuel pellet stack and cladding are modeled with 36 iso-volume ring elements and 8 iso-thickness ring elements, respectively. The axial Linear Heat Rate (LHR) distribution, radial power distribution and coolant inlet conditions are given as input data by the user as burnup history. In this study, neutronic code, RODBURN [14] is used to evaluate radial power distribution history of fuel pellets.

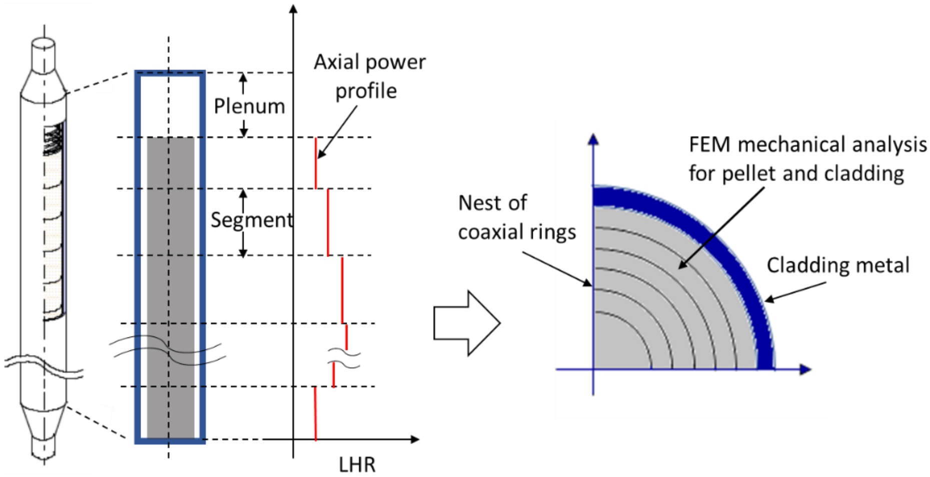

Fig 1. FEMAXI-7 calculation geometry and modeling [14]

2.2 Basic models and fuel pellet property models

FEMAXI-7 incorporates major models for predicting fuel rod performance which have been derived from dedicated separate effect tests. In this study, a set of standard models for UO_2 pellet properties have been selected as was done in the previous study, such as the thermal conductivity, thermal expansion, Young's modulus, Poisson ratio, emissivity, creep, swelling, densification, yield stress, specific heat, and grain growth [12].

For predicting fission gas release (FGR), the standard White-Tucker-Speight model [15] is used to evaluate grain bubble growth and diffusion to the pellet surface. Fission gas atoms are released when the grain boundary gas concentration reaches the FGR threshold limit. The diffusion constant is given by the widely acknowledged Turnbull's model [16]. The pellet-cladding gap conductance is evaluated with a widely used model by Ross and Stoute [17] with considerations of gap gas compositions and temperature, gap size (or PCMI pressure), pellet surface roughness, temperature and hardness.

2.3 Cladding thermos-mechanical property models

(1) Creep model

Since irradiation creep data of the FeCrAl-ODS steel has not been obtained, creep model of stainless steel 304 (SUS304), which is available in FEMAXI-7, is tentatively used for the analyses. The following creep model for SUS304 is used in this study [18].

$$\dot{\varepsilon} = \overline{E}\varphi\overline{\sigma}\{\left(\frac{C_1\overline{E}\varphi}{C_2}\right)\cdot\exp\left(-\frac{\overline{E}\varphi t}{C_2}\right) + C_3\} \tag{1}$$

where, $\dot{\varepsilon}$: creep strain rate (/hour), $\overline{E}$: neutron energy (MeV), φ : fast neutron flux (n/cm^2-s), $\overline{\sigma}$: equivalent stress (psi), t : time (hour), C_1=1.7$\times 10^{-23}$, C_2=5.5$\times 10^{15}$, C_3 =2.7 $\times 10^{-26}$. As shown in Eq. (1), this creep model neglects thermal creep of stainless steel, which may be reasonable under normal operation condition.

Figure 2 compares creep strain rate of SUS304 model with that of a typical Zircaloy model, which is based on the Halden irradiation data [19], under simulated BWR operation condition. As can be seen, creep strain rate of SUS304 is about an order of magnitude smaller than that of Zircaloy, which may have significant influence on the fuel performance during normal operation.

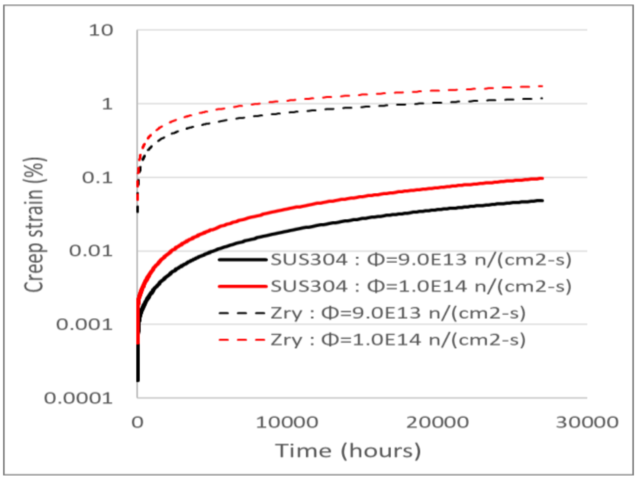

Fig 2. Comparison of irradiation creep rate between SUS304 and Zry

(2) Other models

For some other thermo-mechanical property models (thermal conductivity, coefficient of thermal expansion, Young's modulus, Poisson's ratio, density, specific heat), those of 12Cr-ODS steel are used as a best-estimate substitute to those of FeCrAl-ODS since the compositions of these materials are similar with expectation of similar mechanical properties [20].

3. Reference design and analysis conditions

In this study, a typical BWR 9 by 9 fuel rod design with cladding outer diameter of 11.2 mm is considered as a reference design. However, the cladding wall thickness is assumed to be 0.35 mm by referring to preceding studies, which investigated reactivity and thermal characteristics of the core and the fuel rod [11, 12]. A standard UO_2 pellet with density of 97 %TD is assumed for all cases.

For the axial configuration, a short segment test fuel rod as shown in **Fig. 3** is assumed by referring to typical Halden reactor test irradiations. The fuel stack is divided into 5 calculation segments with different LHR, which assumes that the axial core power distribution does not fluctuate during irradiations. About half of the fuel stack is assumed to consist of hollow pellets, which are for inserting thermocouples to measure the pellet centerline temperature online. A plenum volume of 2.24 cm^3 is assumed, which is about 23 % of the fuel stack volume.

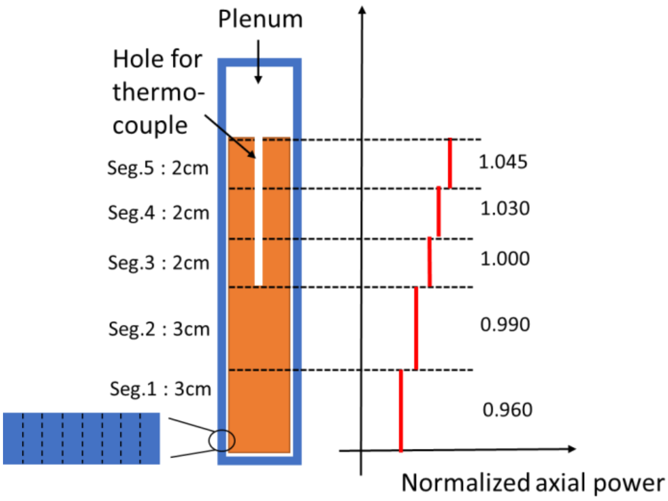

Fig 3. Segment division of test fuel rod

Within each axial segment, the cladding wall is represented by 8 iso-thickness ring elements as shown in **Fig. 3**. In the meantime, the pellet is modeled by 36 iso-volume ring elements and the relative power distribution is given as a function of irradiation time as shown in **Fig. 4**, which has been evaluated by RODBURN under the heavy water cooled Halden reactor condition. For hollow pellet, the power density is reduced to account for the hole. Hence, the peak pellet temperature is expected in Segment 2.

The coolant inlet temperature and pressure are assumed to be 561 K and 9.0 MPa, respectively, which has been tentatively determined from the viewpoint of avoiding bulk boiling in the test section. The Fast neutron flux is tentatively decided on the assumption that it is in about 1.00×10^{12} $(n/cm^2/s)$ / (kW/m). A constant irradiation power history is assumed for an operation cycle length of 3.5 month up to 7 cycles of operation with 6 hours of startup and 6 hours of shutdown with linear power changes between each cycle.

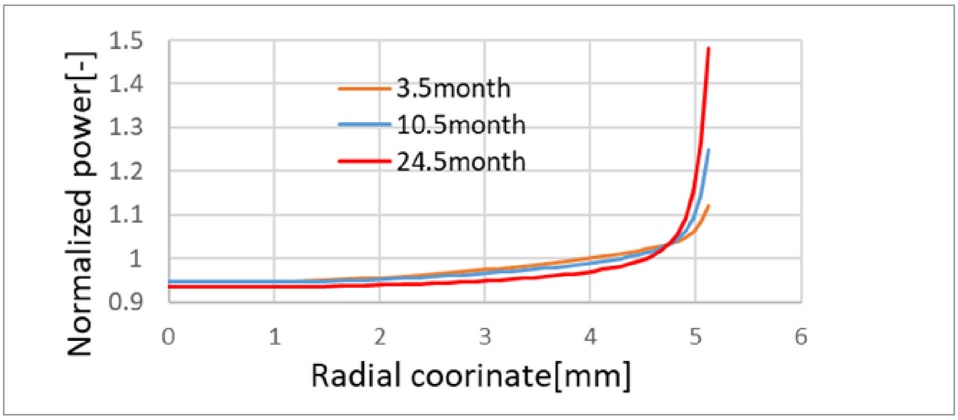

Fig 4. Radial power distribution calculated by RODBURN

4. The irradiation matrix proposal

Considering the features of FeCrAl-ODS steel cladded fuel rod, three test cases have been identified. Namely, (1) Design Target Case, (2) High PCMI case, (3) Cladding Creep-Down Case. Table 1 summarizes proposed design parameters for determination of these three cases.

Case (1) is intended to represent a condition, which corresponds to typical BWR 9 by 9 fuel rod irradiation condition. Case (2) is intended to increase PCMI pressure by increasing LHR to 40 kW/m and replacing He with Ar, which has low thermal conductivity. These changes should increase the pellet temperature and burnup, which should lead to larger thermal expansion and swelling. The initial gap size between the pellet and cladding has also been reduced by half by increasing the pellet diameter. Case (3) is intended to investigate cladding creep-down due to the difference between rod internal pressure and external pressure (coolant pressure). For this purpose, the filler gas initial pressure has been reduced to 0.2 MPa. The gap size is also increased by reducing the pellet diameter. Moreover, these three cases are also designed to investigate difference on the gap heat transfer.

Case #	LHR (kW/m)	Diameter gap (mm)	Filler gas	Filler gas pressure (MPa)	Remarks
(1)	30	0.20	He	1.0	Design target
(2)	40	0.10	95%Ar, 5%He	1.0	High PCMI
(3)	30	0.40	He	0.2	Cladding creep down

Table 1: Irradiation matrix proposal

5. Analysis results

In this Chapter, analysis results are presented by focusing on Segment 2, which is the segment with the highest pellet temperature as described in Chapter 3.

5.1 Thermal behavior

Fig.5 shows the pellet centerline temperature of the three cases. For the design target case, it keeps almost constant at about 1,400 K during the entire irradiation period. In the high PCMI case, the initial pellet centerline temperature is about 270 K higher than the design target case due to higher LHR (40 kW/m) and lower gap conductance (95% Ar filler gas). The gap is closed near the end of the 1st cycle and the temperature steadily rises with irradiation from the 2nd cycle. This is due to the increase in the contact thermal resistance of the pellets and the cladding, because of the release of the Fission Product (FP) gas. The initial temperature of the cladding creep-down case is about 250 K higher than that of the design target case. This is primarily due to lower gap conductance with the larger gap size (0.40 mm). The gap is kept open during the entire irradiation and the temperature significantly increases from the 2nd to the 3rd cycle due to large amount of FP gas release. The maximum temperature is about 2,200 K, which is sufficiently below the melting point. In the meantime, the peak pellet temperature of the hollow pellet in Segment 3 is about 120 K lower than that of Segment 2. Thus, the proposed three cases are expected to show clearly different behaviors, which can be captured through online thermocouple measurements from the 1st cycle.

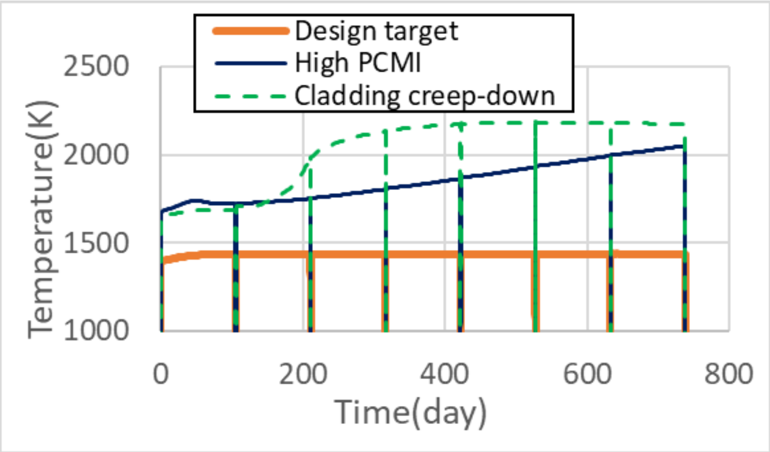

Fig.5 Pellet centerline temperature

Fig.6 and **Fig.7** show fission gas release (FGR) rate and rod internal pressure, respectively for the three cases. The initial low internal pressure of the cladding creep-down case is due to the low filler gas pressure (0.2 MPa). For all cases, the internal pressure does not show much changes in the 1st cycle, but distinctive increase in the pressure can be expected for the High PCMI and cladding creep-down cases from the end of the 2nd / beginning of the 3rd cycle due to significant FGR. Thus, the online internal pressure measurements are expected to provide valuable information to feature the three cases from the end of the 2nd cycle.

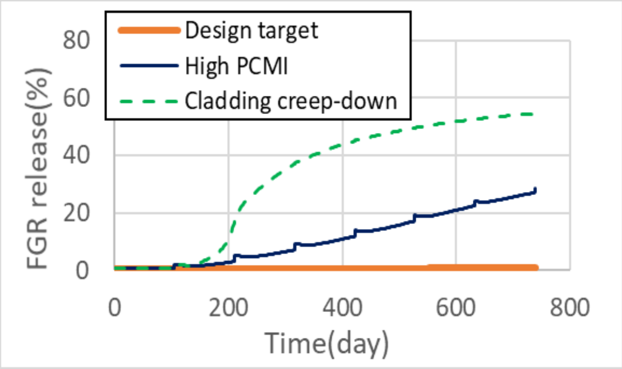

Fig.6 Fission gas release rate

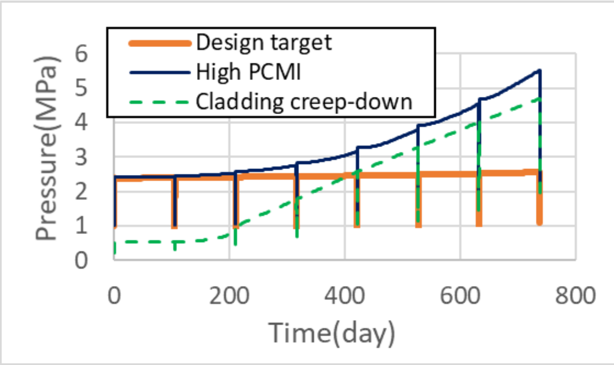

Fig.7 Rod internal pressure

5.2 Mechanical behavior

Fig.8 shows cladding outer diameter changes of the 3 cases. For both the design target and cladding creep-down cases, small creep-down (less than 1 μm) are simulated in the first few cycles. What appears as slight creep-out for the cladding creep-down case after the 4th cycle is actually not creep-out. It is elastic deformation due to change in the internal pressure of the fuel rod. Thus, creep-down of the cladding may not be detected through online measurements, if the irradiation creep characteristics of FeCrAl-ODS steel is similar to that of SUS304. In the high PCMI case, the gap is closed during the 1st cycle. After that, due to swelling of the pellets and PCMI, significant creep-out of the cladding is simulated. The increase of the cladding outer diameter at the end of the 4th cycle is about 10 μm. Thus, a notable difference is expected for the High PCMI case from the 2nd cycle through online measurements of the cladding outer diameter.

Fig.9 shows cladding axial elongation for the three cases. In the simulation, the cladding is assumed to be isotropic and axial elongation due to irradiation (as observed for Zircaloy cladding) is not considered. As the result, cladding elongation due to thermal expansion is dominant. The small fluctuations within each irradiation cycle is elastic deformation, which is corresponding to the radial deformation due to change in the rod internal pressure. In the High PCMI case, gradual and long-term decrease in the cladding elongation across irradiation cycles corresponds to the creep-out in the radial direction. Thus, the online elongation measurements of the cladding elongation may also provide valuable information regarding PCMI.

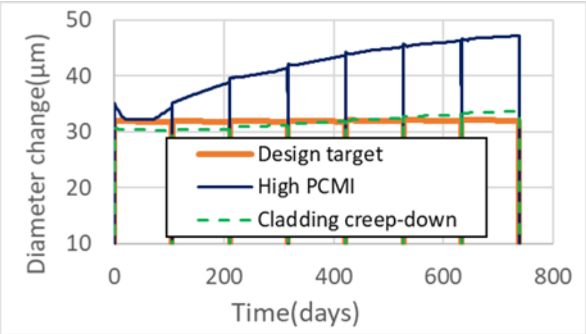

Fig.8 Cladding outer diameter change

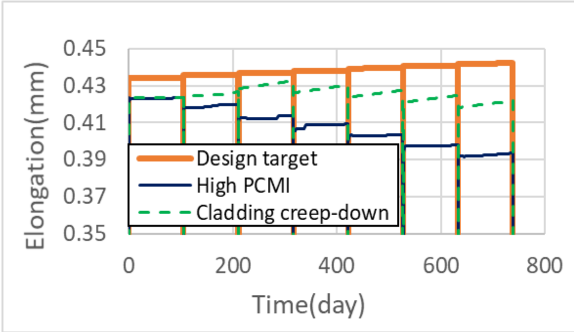

Fig.9 Cladding axial elongation

6. Conclusions

The key parameters and conditions for planning irradiation test for FeCrAl-ODS steel cladded fuel rod have been clarified through fuel performance analyses using FEMAXI-7. The key test cases are the Design Target Case, the High PCMI Case, and the Cladding Creep-Down Case. The key design parameters are linear heat rate, gap size, filler gas compositions and pressure. The differences among the three cases are expected to be well captured through online measurements of the pellet centerline temperature, rod internal pressure, cladding outer diameter change and elongation. However, significant cladding creep down may not be captured under BWR normal operation condition, if the irradiation creep characteristics of FeCrAl-ODS steel is comparable to that of SUS304.

7. Acknowledgement

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